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ITER PHYSICS R&D DURING THE EDA

by Dr. M. Shimada, Head of the ITER EDA JCT Physics Unit

Objective, Structure and Evolution of ITER Expert Group Activity

Physics research and development constitutes the basis for ITER design work. The projections of ITER performance must rest on theory and experimental results from the leading laboratories and universities that together pursue the Parties' fusion science programmes. The main objectives are to strengthen further the physics basis for the inductive Q = 10 operating scenario and to explore scenarios for new modes of operation that could be used to approach steady-state operation.

During the ITER EDA, the Parties have undertaken their physics work for ITER on a voluntary basis outside the framework of task assignments established for ITER tasks in technology R&D and design. Nonetheless the Parties' various efforts are undertaken in a structure designed to offer coherence and co-ordination of the voluntary contributions. The ITER Physics Committee, comprising the Director and the Parties' designated persons for ITER Physics, exercises oversight and is supported by seven Physics Expert Groups (see the article on ITER Physics R&D, IAEA EDA Newsletter Vol. 3, No. 9, and sections on physics progress in the Director's status reports).

With the wholehearted support and commitment of individuals and organizations throughout the Parties, this structure has proved to be extremely effective in providing the necessary physics support to the ITER design activities, the results of which were published as "The ITER Physics Basis" in a special edition of the journal Nuclear Fusion (December 1999). At the same time, ITER has proved a catalyst to general progress in tokamak Physics through the discipline and focus required to identify and address efficiently the main challenges of establishing the ITER Physics Basis. After the publication of the ITER Physics Basis, the results obtained from the continued physics R&D have produced a large number of publications and have been reflected in the physics analysis of ITER.

The seven ITER Physics Expert Groups have been very successful in investigating the area specific to each group and also interdisciplinary areas. Expert Group Meetings, held twice a year on average, are often held jointly with other groups to cope with interdisciplinary issues. Additional experts are also invited for in-depth discussions on specific issues. Furthermore, the arrangements for continued interaction with US fusion scientists on generic issues of tokamak physics have proceeded smoothly, through the invitation of US scientists to an International Meeting, organized before an Expert Group Meeting.

A new framework for co-operation on physics, called the International Tokamak Physics Activity (ITPA), is being planned to operate after the end of the EDA in accordance with a Statement of Position from the IFRC of the IAEA. The new framework should put in place a structure similar to ITER Expert Groups. This co-operation will continue and broaden the voluntary physics activities beyond the ITER EDA Agreement. The ITPA will aim at co-operation in development of the physics basis of burning tokamak plasmas, including ITER; construction, management and updating of databases; and development of scaling and modelling to enhance worldwide tokamak research progress. The full participation of US scientists in ITPA will enhance the progress of physics research.

Progress in Physics R&D

Confinement Database and Modelling

The updated ELMy H-mode confinement database, which now contains gas-puff injection data and new isotope data, including D-T data from JET, still fits rather well the conventional ELMy H-mode energy confinement scaling IPB98(y,2) and the JCT is advised to continue to use this expression in performance studies. The confinement time confidence interval has been revised to $\pm$13%.

Recent experiments in JET and ASDEX-Upgrade show that the value $H_4 \sim 1$ of the enhancement factor of the energy confinement time given by the IPB98(y,2) scaling can be attained at high densities close to the Greenwald density, with high triangularity ($\delta \sim 0.5$) at $q_{95} \sim 3$. In JET, high normalized beta ($\beta_N \sim 2$) and low $Z_{eff} \sim 1.5$ have been achieved simultaneously with the high confinement at high density. This result strongly supports the scenario of achieving Q = 10 in inductive operation of ITER. Good confinement at high densities has also been obtained with high-field-side pellet injection (JET, ASDEX-Upgrade, JT-60U).

The profile database is again operational, permitting access through the website. The Nuclear Fusion paper on the structure of the database has been published.

Transport and Internal Barrier Physics

As suggested by the numerical simulations for Ion Temperature Gradient (ITG) or Electron Temperature Gradient (ETG) turbulence, evidence of a critical temperature gradient is shown in DIII-D, ASDEX-U and Tore Supra. Particularly in ASDEX-U, stiffness of the temperature profile is also seen.

Theory-based transport models (Weiland and GLF23 models) are successful in reproducing experimental profiles of core plasma temperature. Development of an integrated transport model for core, pedestal and divertor plasma is also in progress. All these models support the attainment of Q > 10 in inductive operation of ITER, under the condition of high pedestal temperature ($\geq$4 keV) due to an efficient transport barrier at the separatrix (H-mode). Study of the conditions for the occurrence and characterization of an Internal Transport Barrier (ITB) is in progress; an ITB database has been started, but much more work is needed for a valid projection for ITER.

MHD, Disruptions and Control

The complete stabilization of the 3/2 Neoclassical Tearing Mode (NTM) and the subsequent recovery of beta were demonstrated in ASDEX-U, JT-60U, DIII-D and COMPASS (2/1 mode) at low current drive powers.

On DIII-D, a set of six saddle coils is installed in the midplane. With the use of in-vessel poloidal field sensors for feedback, it was possible to stabilize for almost a second the Resistive Wall Mode (RWM) under normalized betas approaching twice the no-wall limit. A simple extrapolation of DIII-D results to ITER suggests that RWM control is possible within the capability of the saddle coils and power supplies planned in ITER.

As for disruptions, in ASDEX-U a neural network predicting the time till disruption is now routinely in real-time operation, triggering avoidance and mitigation (killer pellets) measures. Runaway electrons are rarely observed in divertor experiments having Vertical Displacement Events (VDEs) during disruptions. Should they occur, they are eliminated by helical instabilities at $q \approx 2$ (JT-60U).

Edge and Pedestal Physics

ELMs with small amplitudes are observed in JT-60U, ASDEX-U and Alcator C-Mod with triangularity higher than 0.4 and safety factor (q_{95}) higher than 3.5-4. This suggests that regimes of small ELMs are accessible in ITER for hybrid and non-inductive operation.

A collaborative work between JT-60U and DIII-D suggests that type I ELMs are triggered by intermediate n (~5-8) ideal MHD mode and type II ELMs, are triggered by higher n (>10), thus affecting a narrower radial region. Multimachine analysis shows a favourable effect of triangularity for higher pedestal pressure and consequently higher confinement. A simple model is proposed for the pedestal, based on turbulence suppression by ExB shearing rate and magnetic shear. An initial check against measurements in JET and Alacator C-Mod looks promising. Projection to ITER based on this model suggests that high pedestal temperatures (3.5 keV) can be attained, which is required for a good core confinement according to the ITG turbulence model.

SOL and Divertor Physics
Multimachine analysis of ELM energy, deposition and reduction has progressed. The predicted ELM energy is ~12 MJ in a reference operation in ITER, albeit with a large uncertainty in projection, raising concern about the ablation of divertor plate material. Inclined target plates made of tungsten would withstand the ELM heat load, but this option should be taken after confirmation of the high ELM heat load and reduction of the disruption heat load.

More measurements in the SOL and divertor revealed transport characteristics (drift, diffusion and wall neutral source) not included in the simulation codes. The electron density profiles are significantly different between JET and ASDEX-Upgrade, which produced different performance in divertor radiation. The chemical sputtering data recently measured in JT-60U and JET show little dependence on particle flux, in contrast to the previous results of TEXTOR.

Diagnostics
In the diagnostics area, key issues which might arise both in plasma control and in physics analysis of ITER have been identified and means of resolving them are being studied.

- Two techniques for measuring the q-profile are being pursued: polarimetry and motional Stark effect spectroscopy.
- Work has continued on the plasma facing mirrors. An extensive, co-ordinated three Party experimental programme has shown that mirrors made from monocrystalline refractory metals can have a long lifetime even at charge exchange flux levels of the order of those expected at the first wall.
- The measurement requirements in the divertor region are being critically reassessed in collaboration with the Divertor Expert Group.

Energetic Particles, Heating and Steady State Operation
- Ripple losses calculated in advanced scenarios with low current density near the axis are negligible in ITER with ferritic inserts, which reduce the toroidal field ripple.
- Experiments in JFT-2M have demonstrated the effect of ferritic inserts in ripple loss reduction.
- Efficient current drive is observed with NNB (JT-60U) and full current drive has been achieved by ECCD in TCV.
- An ITER design type ICRH antenna is planned to be tested in JET.
- Improved coupling is observed for LHCD in H-modes + ITB (JET).
- ECCD is shown to be efficient in stabilizing the NTM. DC is as efficient as AC. The power required is in agreement with a simple theory. For ITER, ECCD power of 20 MW appears to be adequate.
- Promising steady state scenarios (e.g. q_{95} ~ 4-5, $H_{H(y,2)}$ ~ 1.6, H_{89} ~ 2.5, β_N ~ 2.5, q_{min} ~ 1.5-2.0, I_{BS}/I_p ~ 50%) are demonstrated in JT-60U, JET, DIII-D and ASDEX-Upgrade, but substantial development is still needed (active profile control, fuelling, divertor compatibility, RWM control).

Expert Group Structure

Expert Group	Area of Activity
Diagnostics	Physics of plasma measurements and principles of their integration into the design relative to anticipated requirements
Edge and Pedestal Physics	Physics processes in the edge plasma inside the separatrix. H-mode transition physics. Transport and ELMs in the barrier region.
MHD, Disruptions and Control	Global plasma processes and their consequences, MHD instabilities, physics of ELMs, operational limits and their associated design requirements. Control methods to improve plasma performance.
Energetic Particles, Heating and Steady-State Operation	Effects of energetic particles created by auxiliary power sources or thermonuclear reactions on the evolution and stability of plasma processes. Auxiliary power methods to create energetic particles, plasma heating, current drive and rotation. Profile control for steady-state operation

Scrape-off Layer and Divertor Physics	Plasma physics and plasma-material interactions in the scrape-off layer plasma. Control of heat flux onto material surfaces as well as control of particle inventory by pumping.
Transport and Internal Barrier Physics	Core turbulent transport of heat, particles and angular momentum and its control as exemplified by internal and H-mode transport barrier (theory and experiment).
Confinement Database and Modelling	Assembly of global and profile databases for nominal ELMy H-mode operation as well as advanced/ITB scenarios. Development, archiving and testing of empirical and first-principles transport models for these scenarios.

Members of ITER Physics Committee and Expert Groups (July 2001)

	Chair	Co-Chair	EU	JA	RF
Physics Committee	R. Aymar	M. Shimada			
Designated Persons			D. Campbell	M. Wakatani H. Ninomiya	N. Ivanov
Senior Physicists			K. Lackner	T. Tamano K. Ushigusa	Y. Dnestrovski S. Mirnov
Diagnostics	A.J. Donne	A. Costley	F. Orsitto	H. Zushi Y. Kusama T. Sugie	V. Strelkov A. Kisliakov A. Krasilnikov
Scrape-off-Layer and Divertor Physics	N. Asakura	A. Loarte	G. Matthews J. Neuhauser V. Phillips P. Ghendrih	T. Kato S. Takamura A. Sakasai	V. Kurnaev
Edge and Pedestal Physics	Y. Kamada	G. Janeschitz	L. Horton W. Suttrop H. Weisen	N. Ohyabu T. Hatae	V. Osipenko
MHD, Disruption and Control	O. Gruber	Y. Gribov	O. Gruber T. Hender J. Lister	S. Tsuji-Iio T. Ozeki S. Takeji	V. Lukash S. Mirnov N. Ivanov
Energetic Part-icles,Heating and Steady State Operation	C. Gormezano	K. Miyamoto	A. Becoulet A. Jaun	A. Fukuyama Y. Takase K. Tobita S. Ide	K. Razumova V. Vdovin S. Konovalov
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Confinement, Database and Modelling	G. Cordey	A. Polevoy	F. Ryter J. Connor	T. Takizuka Y. Ogawa Y. Miura	V. Leonov

* Shading indicates Physics Committee Members

ITER TECHNOLOGY R&D DURING THE EDA

by T. Mizoguchi, ITER EDA Joint Central Team, Naka JWS

In the past nine years of the ITER EDA, the good collaboration between the Joint Central Team and the Home Teams' many institutes, industries and universities has led to a comprehensive and coherent technology R&D programme to qualify the ITER design. About 800 R&D Task Agreements were developed and agreed between the Director and the Home Team Leaders. The total R&D resources committed in nine years by the Parties amounted to about 660 kIUA (1 kIUA = 1 M$ in 1989). These R&D resources were distributed to the various technology R&D areas as shown in the table below. Three quarters of the resources have been devoted to the areas that include the seven large R&D projects, qualifying all the major key components of the basic machine of ITER and their maintenance tools.

Percentage of Resources Devoted to the Different R&D Areas

R&D Area	**%**
Magnets (incl. L-1 CSMC & L-2 TFMC Projects)	27.9
Vacuum Vessel (incl. L-3 VV Sector Project)	5.3
Blanket and First Wall including Materials (incl. L-4 Blanket Module Project)	16.3
Divertor & PFC including HHF Materials (incl. L-5 Divertor Cassette Projects	15.1
In-vessel Remote Handling (L-6 Blanket and L-7 Divertor Remote Handling Projects)	11.3
Subtotal	75.8%
Fuelling & Pumping	1.9
Tritium System	3.4
Power Supply	1.8
IC H&CD	1.1
EC H&CD	3.7
NB H&CD	3.1
Diagnostics	2.5
Safety Related R&D	3.4
Miscellaneous (incl. Standard Component Development)	3.3
Total	100.0

Total Resources: 660 kIUA (1 kIUA = 1 M$ in 1989). Contributions from the EU, JA, RF and US Parties are about 1/3, 1/3, 1/6 and 1/6, respectively. The US contributed only up to July 1999.

Major developments, fabrication and tests have been completed. The technical output from the R&D validates the technologies incorporated in the ITER design, confirms the manufacturing techniques and quality assurance, and supports the manufacturing cost estimates for important cost drivers. The testing of models demonstrates their performance margins and/or optimizes their operational performance.

Valuable and relevant experience has already been gained in the management of industrial scale, cross-Party ventures. The success of these projects increases confidence in the possibility of jointly constructing ITER in an international framework.

The overall review of the ITER Technology R&D has been published in July 2001 in the journal "Fusion Engineering and Design". A short overview of the ITER Technology R&D achievements is presented below.

The R&D programme in the area of **Superconducting Magnets** (Projects L-1 and L-2) consists of the development of the superconducting magnet technology to a level that will allow the various ITER magnets to be built with confidence. It drives the development of a conductor capable of very high current and field, including the manufacturing of strand, cable, conduit and terminations, and the conductor R&D in relation to AC losses, stability and joint performance. A total of 29 t of Nb_3Sn strand, from seven different suppliers throughout the four Parties, was produced and qualified. This reliable production expanded and demonstrated the industrial manufacturing capability for the eventual production of the 480 t of high performance Nb_3Sn strand as required for ITER. The R&D programme covers coil manufacturing technologies, including electrical insulation, winding processes (wind, react and transfer) and quality assurance.

L-1 Central Solenoid Model Coil

For the central solenoid model coil, the next critical step, the heat treatment to react the superconducting alloy without degrading the mechanical properties of the Incoloy jacket, was successfully completed by having a very low oxygen content in the oven atmosphere, and low tensile stresses in the jacket. By using approximately 25 t of the strand, the inner module (US), the outer module (JA) and the CS insert coil (JA) were fabricated, then assembled in the ITER dedicated test facility at JAERI. In April 2000, the maximum field of 13 T with a cable current of 46 kA was successfully achieved. The stored energy of 640 MJ at 13 T was safely dumped with a time constant as short as 6 s. By comparison, the energy discharge time in the full size ITER CS is 11 s. The CS insert coil was successfully tested with 10 000 cycles by August 2000 to simulate ITER operation (0 to 40 kA cycles in a steady 13 T background field of the CS model coil). The size of the CS model coil (3.6 m in diameter and 2 m in height) is almost the same as that of one of the six modules (4 m in diameter and 2 m in height) of the ITER central solenoid, and the maximum field and the coil current are the same. The CS model coil is the largest high field, pulsed superconducting magnet in the world.

L-2 Toroidal Field Model Coil

The TF model coil was fabricated in the EU. It has a cable similar to the full-size TF coil cable of circular cross-section and a thin stainless steel conduit. The diameter of the TF model coil is smaller but the cross-section is comparable in size to that of the ITER TF coil. After being assembled in the test facility at Forschungszentrum Karlsruhe (FKZ), the TF Model Coil was ramped up in July 2001 to the maximum current allowed by the facility of 80 kA, corresponding to a maximum field of nearly 8 T. By comparison, the peak field and the operating current are 11.8 T and 68 kA in ITER. The behaviour of the superconductor and the joint resistance are in line with expectations. The NbTi busbars carried 80 kA at a maximum field of about 3 T at 5 K, and the joints at that field level behaved better than expected. The test programme will continue to explore the operational limits of the coil and validate the design codes in particular by assembling the TF Model Coil inside the bore of one of the large LCT coils to explore higher field values and out of plane loads. In addition, a TF insert coil with a single layer is now being tested inside the bore of the CS model coil test facility at JAERI at a field of up to 13 T. This insert coil had been completed in the RF in early 2001 and was assembled inside the bore of the CS Model Coil in August 2001.

For the development of the manufacture of the TF coil case, large forged and cast pieces (about 30 t and 20 t, respectively) have been produced in the EU. Investigation of the properties of the forging has revealed values exceeding the requirements of 1000 MPa yield stress and 200 MPa $m^{1/2}$ fracture toughness, with low fatigue crack growth rates. The casting also shows properties adequate for the low stress regions of the case (yield stress about 750 MPa). Welding trials have demonstrated successful welding of the cast to forged sections when the level of manganese is high enough in the steel and/or in the welding filler metal. For the case assembly welds, electron beam welding is planned for the first pass followed by submerged arc welding for the remainder, to minimize distortion. The welding processes have been qualified, but non-destructive testing of welds in cast metal remains a difficulty.

L-3 Vacuum Vessel Sector

In the vacuum vessel sector project, the main objectives were to produce a full-scale sector of the ITER vacuum vessel, including the equatorial extension port, to establish the tolerances, and to undertake initial testing of mechanical and hydraulic performance. The key technologies have been established and, in relation to manufacturing techniques, two full-scale vacuum vessel segments (half-sectors) have been completed by JA industrial firms, using a range of welding techniques, within the required tolerances. At JAERI, the half-sectors were welded to each other and the equatorial port fabricated by the RF was attached to the vessel to simulate the field joint which will be made at the ITER site during the assembly of the machine. Remote welding and cutting systems prepared by the US were also tested and applied.

L-4 Blanket Module

The blanket module project was aimed at producing and testing full-scale modules of the first wall elements and full-scale, partial prototypes of mechanical and hydraulic attachments of the modules to the vacuum vessel, as well as at demonstrating prototype integration in a model sector. The key technologies have been successfully developed, tested and qualified.

- Material interfaces such as Be-Cu and Cu-stainless steel have been successfully bonded by using hot isostatic pressing (HIP) and other advanced techniques.
- In parallel, heat cycle and irradiation tests have been performed for the base materials and the bonded structures, and have demonstrated that the performance in power flux on the armour material (Be) is well within the acceptable level.
- A full-size shield blanket module has been completed by using powder HIP. After the first wall has been attached to this block, the module will be tested.
- A full-scale primary shield blanket module, without Be armour and attachments, has been completed in the JA and demonstrated fabrication feasibility. Destructive examination for manufacturing defects was performed by cutting up the module.
- Flexible mechanical attachments made of titanium alloy have been developed and tested in the RF.

L-5 Divertor Cassette

The divertor cassette project aimed at demonstrating that a divertor can be built within tolerances and withstand high thermal and mechanical loads.

- A full-scale prototype of a half-cassette has been built by the four Parties. Plasma-facing components (PFC) shipped from JA and the RF were installed on the inner divertor cassette body fabricated in the US, and hydraulic flux and mechanical tests were performed at Sandia National Laboratory. Other PFC mock-ups fabricated by the EU and the RF were also installed on an outer divertor cassette body fabricated by the EU.
- Various high heat flux components were fabricated and tested by the four Parties. High heat cycle tests show that carbon fibre composite (CFC) monoblock survives 20 MW/m^2 x 2000 cycles (EU) and tungsten armour survives 15 MW/m^2 x 1000 s (EU/RF). A large divertor target mock-up with CFC attached to dispersion-strengthened copper (DSCu) through oxygen-free copper (OFCu) has been successfully tested with 20 MW/m^2 x 1000 cycles from a large hydrogen ion beam with a diameter of 40 cm (JA).
- High heat flux tests have been also performed on irradiated components. For example, CFC brazed on Cu survived 20 MW/m^2 x 1000 cycles after 0.3 dpa irradiation at 320°C. Tests with pulsed heat deposition simulating the thermal load due to disruptions have demonstrated erosion but no disruptive failure of CFC armours even with 0.4 dpa irradiation. (The average neutron fluence of 0.3 MW/m^2 at the first wall gives 0.38 – 0.59 dpa on the CFC divertor target.)

L-6 Blanket and L-7 Divertor Remote Handling Systems

The last two of the large projects focus on ensuring the availability of appropriate remote handling technologies which allow intervention to repair or exchange damaged modules in contaminated and activated conditions on reasonable timescales. In this area, full-scale tools and facilities have been developed. Their testing will be extended over a long period of time until the ITER operation phase. This is necessary not only for developing the right procedures but also for optimizing their use in detail and minimizing any intervention time. Rescue procedures and equipment to recover other equipment and components in the event of failure during the procedure are being developed. The facilities will also allow training of operators.

The blanket module remote handling project involves proof-of-principle and related tests of remote handling scenarios, including opening and closing the vacuum vessel flange, and the use of a transport vehicle on a monorail inside the vacuum vessel for the installation and removal of blanket modules. At first, the procedures were demonstrated at about ¼ scale to reduce the risk and cost. After successful operation of the ¼-scale model, the fabrication of a blanket test platform with full-scale equipment and tools, which simulates the full-scale structure of a 180° ITER in-vessel region and incorporates a 180° rail, a vehicle with telescopic type manipulator and a welding/cutting/inspection tool, has been completed in JA. The simulation of installation and removal of a simplified, 4 t dummy shield blanket module has been successfully performed by using a teach and playback procedure. The dummy module was installed with only 0.25 mm clearance between dummy keys and keyways using the intrinsic compliance of the manipulator. These integrated tests in this blanket test platform are providing a comprehensive validation of the remote handling system so as to allow completion of the detailed design of the components.

In the divertor remote handling project, the main objective is to demonstrate that the ITER divertor cassette can be installed and removed remotely from the vacuum vessel and remotely refurbished in a hot cell. This involves the design and manufacture of full-scale prototype remote handling equipment and tools and their testing in a divertor test platform to simulate a portion of the divertor area of the tokamak, and the design and manufacture of a divertor refurbishment platform to simulate the refurbishment facility. Construction of the necessary equipment and facilities has been completed in the EU and successful tests have been carried out with the remote handling transporters and tools procured in the EU, as well as a central cassette carrier from JA and a transporter from Canada. The system is based on a toroidal transporter that moves on the rails to which the individual divertor cassettes are attached. The transporter can move a cassette to a remote handling port through which the cassette is extracted by a radial transporter that is deployed from a transfer cask docked to the port. Modification of equipment is under way to take account of the latest design details.

The real in-vessel operation will be carried out in a gamma field of 10^4 Gy/h. Key elements, such as motors, position sensors, wire/cables, glass lenses, electrical insulators, periscopes and strain gauges, have been shown to operate at up to 10^6 –10^7 Gy.

Other R&D

In addition to the seven large R&D projects, development of components for fuelling, pumping, tritium processing, heating/current drive, power supplies and plasma diagnostics, as well as safety-related R&D, have significantly progressed, as exemplified below:

- A tritium pellet injector has been tested with a throughput of 36 g T_2 and 28 g DT, and ejection of a large pellet (10 mm) from an 80 cm radius curved guide tube has been successfully achieved at 285 m/s in the US. Further tritium pellet injector development is being continued in the RF.
- A half-scale cryogenic pump for DT, He and impurities has been completed and is under test in the EU.
- A fully integrated fuel cycle system has been tested in the US using about 100 g of tritium throughput under various operation conditions, including 25 days of around the clock operation.
- Gyrotrons at 170 GHz have been developed and successfully operated in JA at 0.9 MW x 9 s with a chemical vapour deposited diamond window from the EU, and at 1 MW x 1 s in the RF.
- Key components for the ion cyclotron plasma heating antenna and the transmission line have been developed and tested at a higher voltage than the expected operational voltage.
- Almost full-size negative ion beam sources and high voltage acceleration technology (1 MeV) have been developed in JA and the EU for neutral beam plasma heating.
- For the magnet system power supplies, mechanical bypass switches and fast-make switches have been developed and successfully tested at 66 kA, and explosively actuated circuit breakers have been developed and successfully texted at 66 kA and 170 kA at the Efremov Institute.
- Irradiation tests of key components of plasma diagnostics have provided the values required to design shielding of components and to assess the need for replacement. The effect of radiation-induced EMF (RIEMF) on wires of magnetic probe measurements is an unexpected issue and is under study. The lifetime of mirrors positioned near the plasma will be limited by deposition/sputtering and is under investigation.

Safety-related R&D, such as the characterization of dust in tokamaks, tritium co-deposited with carbon, and experiments on steam-material reactions, has provided inputs for key phenomena and data for safety assessments, and the current R&D emphasis is on verification and validation of data, models and computer codes. Neutron shielding tests using 14 MeV neutron sources in JA and the EU demonstrate that the accuracy of activation calculations is within 10%.

EDA ACTIVITIES RELATED TO SAFETY

by Drs. C. Gordon and J. Raeder, Safety, Environment and Health Group, ITER EDA Joint Central Team, Garching JWS

Introduction

This article reviews the accomplishments in ITER safety analysis during the course of the Engineering Design Activities (EDA). The key aspects of ITER safety analysis are:

- effluents and emissions from normal operation, including planned maintenance activities;
- occupational safety for workers at the facility;
- radioactive materials and wastes generated during operation and from decommissioning;
- potential incidents and accidents and the resulting transients.

This work on ITER safety required the integration of detailed analyses by a geographically dispersed safety team consisting of JCT and Home Team experts who initially had different approaches and methods based on conceptual fusion reactor studies and fission power plant practices.

Since an ITER site has not yet been selected, a safety approach was developed for a generic site in such a way that compatibility with the Parties' regulatory frameworks can be expected. The work on safety has contributed to a design providing the basis for regulatory approval with the expectation that only minor changes will be needed to meet the host country's regulations. After siting, safety design and implementation will be finalized in accordance with the host country's regulations and practices.

The implementation of a generic safety approach, the safety aspects of the design, and the assessments of effluents, occupational safety, waste and accidents, are intended to support regulatory submissions by any Party. The assessment is documented in the Generic Site Safety Report (GSSR), whose main purpose, in combination with other ITER documentation, is the provision of this technical information. The GSSR comprises about 1100 pages of text and figures, summarizing the contributions of designers and analysts from the JCT and Home Teams in 11 volumes ordered as follows:

Volume I	Safety Approach
Volume II	Safety Design
Volume III	Radiological and Energy Source Terms
Volume IV	Normal Operation
Volume V	Radioactive Materials, Decommissioning and Waste
Volume VI	Occupational Safety
Volume VII	Analysis of Reference Events
Volume VIII	Ultimate Safety Margins
Volume IX	External Hazards
Volume X	Sequence Analysis
Volume XI	Safety Models and Codes

As a result of the work during the EDA it is concluded that ITER is safe, with little dependence on engineered, dedicated safety systems for public protection because of the fail-safe nature of the fusion energy reaction, limited mobilizable radioactive inventories, multiple layers of confinement, and limited decay heat and passive means for its removal.

Safety Objectives

ITER aims at demonstrating the safety and environmental potential of fusion, thereby providing a good precedent for the safety of future fusion power reactors. Hence, ITER has to address the full range of safety hazards which were taken into account by including extensive safety and environmental assessments in the development work.

The technical safety objectives are broken down into:

General Safety

Protect individuals, society and the environment; ensure in normal operation that exposure to hazards within the premises and due to any release of hazardous material from the premises is controlled, kept below prescribed limits and minimized; prevent accidents with confidence, to ensure that the consequences of more frequent events, if any, are minor; ensure that the consequences of accidents are bounded and that their likelihood is small.

No Evacuation

Demonstrate that, in the light of the favourable safety characteristics of fusion and appropriate safety approaches, it is technically justifiable to assume that the hazard from internal accidents is reduced to a level such that the recommended IAEA criteria for evacuation of the public will not be exceeded.

Waste Reduction

Reduce radioactive waste hazards and volumes.

Safety Principles

The design and the ongoing safety reviews and assessments were guided by the following principles:

As Low As Reasonably Achievable (ALARA)
Exposures to hazards are kept as low as reasonably achievable, economic and social factors taken into account.

Defence-in-Depth
All activities are subject to overlapping levels of safety provisions so that a failure at one level would be compensated by other provisions. Priority is given to preventing accidents. Protection measures are implemented in subsystems as needed to prevent damage to confinement barriers. In addition, measures to mitigate the consequences of postulated accidents are provided, including successive barriers for confinement of hazardous materials.

Passive Safety
Passive safety is given special attention. It is based on natural laws, properties of materials, and internally stored energy. Passive features, in particular minimization of hazardous inventories, help ensure ultimate safety margins.

Consideration of Fusion's Safety Characteristics
The safety approach is driven by a deployment of fusion's favourable safety characteristics to the maximum extent feasible:

- the plasma burn is self-limiting with regard to power excursions, excessive fuelling and excessive additional heating;
- the plasma burn is passively terminated by the ingress of impurities under off-normal conditions;
- the plasma burn is terminated inherently when fuelling is stopped owing to the limited confinement by the plasma of energy and particles;
- the fuel inventory in the plasma is always below 1 g so that the fusion energy content is small;
- the radioactive decay heat density is low;
- the energy inventories are relatively low;
- large heat transfer surfaces are available together with large masses acting as heat sinks;
- confinement barriers exist and must be leak-tight for operational (non-safety) reasons.

The experimental nature of the facility is also addressed. A conservative, fault-tolerant safety envelope is provided to allow flexible experimental usage. In view of the limited operational experience with DT plasmas, experimental components are conservatively designed, taking into account the expected loads from plasma transients so as to reduce the demands on systems which are required for safety. A safety function is not assigned to experimental components, but faults in these are considered as expected events in the safety assessments. The experimental programmes and related machine modifications and operations are developed to take advantage of knowledge gained during preceding operations.

Review and Assessment
Safety assessments are an integral part of the design process and results are used to assist in design improvements and in the preparation of safety documentation for regulatory applications. These analyses comprise normal operation and all categories of off-normal events.

To confirm the acceptance of the generic safety approach, meetings of the Parties' Designated Safety Representatives (i.e. regulators) were convened. A consensus on the safety principles and criteria was reached, based on internationally recognized International Commission on Radiological Protection (ICRP) and IAEA recommendations, in particular on the concept of defence-in-depth and on the ALARA principle. Furthermore, it was agreed that the scope of the implementation and documentation outlined for ITER appears to be a reasonable basis.

Safety and Environmental Criteria

Quantitative safety and environmental criteria have been set against which to judge the success in achieving the safety objectives. These are based on internationally recognized criteria, most notably those recommended by the ICRP and the IAEA.

The quantitative release guidelines are used as a surrogate for regulatory dose limits. They are expressed in physical units (such as grams of tritium oxide released to the environment) rather than doses (expressed in sieverts) since the definitions of doses, the way they are calculated and the related regulatory limits depend on specific sites and vary from Party to Party. The guidelines were derived by considering a range of dispersion parameters, site characteristics and dose definitions so as to be compatible with the more restrictive of the regulatory limits in the Parties. The underlying concept is that events with the greatest likelihood of occurrence should have the least consequences.

An additional category, hypothetical events of such low probability that they can essentially be regarded as an almost impossible combination of failures, is also assessed to demonstrate the ultimate safety margins of the design. The favourable safety characteristics of fusion manifesting themselves in ITER can be further demonstrated if, even for hypothetical events, the calculated doses to the local population are below 50 mSv (early dose). This value is below the generic optimized intervention level for temporary evacuation recommended by the IAEA, which is 50 mSv avertable dose within a period of no more than one week. ICRP and IAEA recommendations were also used to set guidelines for occupational safety.

Safety Approach

The objective in developing the safety approach was to provide confidence to each Party that the design of ITER is safe and licensable, and in so doing to define the facility's safety performance requirements. Early in the EDA, two important issues needed to be addressed, namely, the fact that the safety characteristics of fusion are different from those of nuclear fission and the fact that the safety and regulatory approaches vary from Party to Party. Therefore, the detailed knowledge from conventional nuclear fission power plants can be adopted for ITER only after extensive revision, based on careful review and interpretation. The ‘national’ approaches are internally consistent, but nevertheless it is not reasonable to apply specific points in isolation to ITER without detailed consideration. The solution was to develop, until a site is selected, an ITER-specific safety approach which is characterized by the following:

- It is comprehensive and internally consistent.•
- It uses agreed basic international safety concepts.
- It is tailored to fusion hazards and safety characteristics.
- It is generally compatible with the Parties’ regulatory approaches.

Safety Functions

Protective measures and safety functions were identified by a review of the hazards in ITER and they were assigned to implementing systems and components. This was done by also accounting for beneficial fusion safety features such as inherent plasma burn termination and low nuclear decay heat. The measures and functions are outlined in the following.

- Restriction of inventories, a fundamental preventive measure, is an effective method to control hazards. Of particular importance is its application to tritium and neutron activation products.
- Confinement of hazardous materials is the fundamental safety function. It refers to all types of physical and functional barriers which protect against the unintended mobilization, spread and release of hazardous materials. Since releases can most significantly occur upon failure of barriers, protection of confinement is needed and the following challenges had to be considered:
  - removal of heat to protect against mobilization (by evaporation or melting, for example) of hazardous materials and breach of barriers;
  - control of coolant enthalpy to prevent damage to barriers (from over- or underpressure, for example);
  - control of chemical energy to avoid energy release and pressurization threats to confinement barriers (particularly from potential reactions of plasma-facing materials with steam);
  - control of magnetic energy to avoid damage to confinement barriers in the event of failures (from mechanical impact or electric arcs, for example).

Assessment Tools and Data

At the start of the EDA, the methods for the safety assessments, including the necessary tools and ITER-specific data, needed to be developed. To this end, a safety-related R&D programme with the Home Teams was pursued which up to the end of 1998 was primarily oriented at understanding and investigating ITER-specific safety issues:

- hydrogen isotope behaviour (accumulation, desorption, permeation) in plasma-facing armour materials (beryllium, carbon fibre composites and tungsten);
- formation of erosion products (dust) from plasma-facing materials (characterization, mobilization, transport, removal and monitoring) under conditions simulating ITER modes of operation;
- chemical interaction of plasma-facing materials (especially beryllium) with steam and air;
- activation product volatility;
- corrosion product generation and transport;
- decay heat measurement of plasma-facing component materials;
- transient thermal hydraulic phenomena.

A significant effort was given to the development of analytical tools to undertake the safety analysis. This included the development of new computer codes and the modification of existing ones for fusion applications. By 1998, ITER analysts had a set of tools capable of analysing

- transient behaviours of plasma-wall interactions,
- thermo-hydraulic behaviour of the heat transport system under anomalous conditions including discharge into vacuum and onto cryogenic surfaces,
- temperature and pressure dependent chemical reactions due to ingress of coolant into the vacuum vessel,
- aerosol, dust and tritium mobilization and transport.

Since 1998, the safety-related R&D has had the main objective of verification and validation of data, computer codes and assessment models.

Results of the Safety Assessments

The assessments were conducted by JCT and Home Team analysts. The need to ensure an integrated and consistent assessment of an evolving design led to the introduction of
- a single safety analysis data list to be used by all analysts,
- a single list of detailed analysis specifications (setting the assumptions and conditions to be analysed) to be followed,

- templates for individual analysis documentation to allow ready assimilation into the overall documentation.

Effluents
To estimate effluents, a systematic approach based on ITER’s Work Breakdown Structure was developed so that each system could be examined as a possible source of effluents. Conservative estimates of expected end-of-life conditions were used in order not to underestimate potential effluents.

The ITER design incorporates many features to ensure that the environmental impact during normal operation will be low, including confinement barriers to prevent releases, and air and water detritiation and filtration systems to treat releases. The hazards are known and the control technologies are well established at existing facilities. Sources of potential effluents were identified, pathways determined, and design features and release control systems were assessed and, under ALARA, ways to reduce the main contributors were examined. Annual effluents will increase with time from the start of operation but will only gradually approach the estimated levels.

Potential doses to a member of the public from ITER operation will be site dependent. For a 'generic' site, the calculated doses would be less than 1% of the natural background level. The continued application of the process to implement the ALARA principle may further reduce the estimated normal releases.

Occupational Safety
For occupational safety, maintenance procedures and human resources estimates developed by the design engineers were used together with estimates of dose rates from external (including deposits of activated corrosion products) and internal (from tritium in air) radiation to estimate occupational exposure, and most importantly to examine the potential for dose reduction.

ITER has established a programme for personnel protection against hazards anticipated during construction, operation and maintenance activities. The highest exposure and risk areas were identified for potential design improvements (shielding, contamination control, reduced operator exposure times, etc.) aimed at reducing overall exposures and ensuring good contamination control. In addition, there will be several years of operation for further learning and optimization prior to the introduction of tritium. The current assessment of all major systems points to ITER successfully maintaining occupational exposures below the project guidelines,which are in line with guidelines for next generation fission power plants.

Waste and Decommissioning
The issue of radioactive materials, decommissioning and waste is being carefully considered. For waste characterization, activation calculations based on a one-dimensional radial build-up of ITER were used, under a conservative assumption for the total neutron fluence.

In the absence of an actual site for ITER, the ultimate waste amounts were estimated provisionally on the basis of ‘clearance'. According to the IAEA concept, materials with activation levels less than the specified clearance level can be released from regulatory control (i.e. control is removed), irrespective of how and where they may be used in the future. The radioactive materials arising during operation and remaining after final shutdown include activated materials (due to fusion neutrons) and contaminated materials (due to activated tokamak dust, activated corrosion products and tritium) and mixtures thereof. Decay and decontamination will reduce the radioactivity with time after final shutdown. Therefore, not all radioactive materials may need to go into waste repositories; rather a significant fraction has the potential to be ‘cleared’. Since this fraction increases with time, the project provisionally assumes that radioactive material not suitable for clearance after a decay time of up to 100 years is ‘waste’ needing disposal in a repository.

The design approach includes provisions for reducing the quantities and hazards of radioactive materials, for example through the use of modular components, appropriate choice of materials, control of impurities, shielding and reuseable components. To ensure that ITER can be safely dismantled at the end of its useful operating life, decommissioning plans have been developed.

The estimated masses are:

- total radioactive material at shutdown — about 30 000 t
- material remaining as waste after a decay time of up to 100 years — about 6000 t.

Accidents

The greatest challenge during the early part of the EDA was to characterize the safety of ITER by analysing in detail the transients resulting from failures. To ensure to the extent possible that all aspects of plant operation have been considered, two fundamentally different approaches have been applied to the identification of initiating events, namely a 'bottom-up' approach (starting upwards from the component level) and a 'top-down' approach (starting downwards from the ultimate consequences of faults). The former systematically catalogues all potential faults in systems and components and considers the conceivable consequences of these faults. The focus is on the failure of individual components and is based on the design in as much detail as is available. In contrast, the top-down approach starts at the plant level and takes a global view of the potential ultimate consequences. By considering the abnormal events which would have to occur to produce these consequences, again a list of event initiators is developed in terms of system or component faults.

The catalogue of events and the event sequences resulting from these studies identified:

- the radioactive inventories at risk,
- the confinement barriers challenged,
- the mitigating systems that must fail for a hazardous plant state to occur,
- the release pathway.

The catalogue has been evaluated to ensure that each event is either clearly insignificant or is covered by a detailed analysis.

A limited set of Reference Events (25 events) was developed consisting of a postulated initiating event and all consequential failures and assumed aggravating failures (additional independent failures in mitigating systems, for example). The Reference Events cover the major systems, the radioactive inventories distributed amongst these systems and the initiator types that have the potential to cause releases. The adequacy of the Reference Events set was confirmed by the detailed event identification described above. It was necessary to proceed in parallel:

- defining and analysing the set of Reference Events (identified by a functional assessment of the facility)
- systematically identifying initiating events on the basis of detailed assessments.

Ultimately it was shown that the consequences of all identified sequences are bounded by the assessed consequences in one or more of the analyses of Reference Events.

In the spirit of conservatism, limiting or bounding conditions were assumed to maximize consequences.

- The plasma behaviour was addressed in a conservative way to show the limited effects of loss of plasma control or exceptional plasma behaviour.
- Loss of power was investigated to determine if there are requirements for the supply of emergency power.
- Many events were grouped around the cooling water systems, which are key features whose safety has to be demonstrated.
- Air and water ingress into the vacuum vessel and cryostat under various off-normal plant conditions was investigated.
- Events during maintenance of the vacuum vessel were considered since maintenance will be a typical state of operation.
- The safety of the tritium plant with its significant inventory was addressed.
- Magnet system structural integrity and the consequences of arcs were examined;
- The consequences of failures of confinement and decay heat removal in the hot cell were investigated.

The analysis of the 1998 ITER design allowed identification of improvements in the further safety design of ITER:

- elimination of the 100 m stack as a requirement for ensuring that off-site dose limits are met;
- changes to confinement design to eliminate failure combinations that lead to bypassing the confinement;
- introduction of a tokamak vent system to help ensure that pressures in the vacuum vessel remain below the pressures in adjacent rooms following accidents;
- improvements to the vacuum vessel pressure suppression system so that pressures following small in-vessel leaks remain below atmospheric pressure.

The results show that radioactive releases for all the Reference Events are well below the project release guidelines, which would lead to doses comparable to the average annual natural background exposure for a generic site. The assessments provide confidence that the operation of ITER will result in no significant risk to the general public from accidents.

In addition, the ultimate safety margins of the facility were examined by analysis of hypothetical events, arbitrarily assuming that more and more failures occur. As a result of the safety characteristics of fusion and engineered design features, it requires an almost inconceivable combination of failures to lead to a significant release of radioactive material. Analysis shows that the design is tolerant to failures, that there is no single component whose failure leads to very large consequences, that there is no single event that can simultaneously damage the multiple confinement barriers, and hence that the design provides a high level of public protection even for these hypothetical events.

Conclusions

Looking back at the activities related to safety carried out during the EDA, a significant step forward for fusion safety has been accomplished. The discipline of an integrated, detailed design and the anticipation of a future regulatory review led to the development and validation of new approaches, tools and data to examine ITER safety. The need to integrate the Home Team and JCT analyses in a single assessment led to the development of means of ensuring a cohesive and internally consistent document. The resulting safety analyses have led to the most comprehensive and well-documented assessment of a fusion design to date.

In summary, the assessments documented in the GSSR show that ITER can be constructed and operated safely and without significant environmental impacts:

- Effluents during normal operation are estimated for a generic site to lead to doses to the public which are less than 1% of natural background radiation levels.
- Occupational exposure of workers is estimated to be within guidelines set for next generation nuclear power plants.
- The majority of the radioactive materials from operation and decommissioning will not remain as waste beyond a human timescale if the present concept of clearance can be applied.
- Doses from accidents are estimated to lead to doses to the population that are at worst comparable to the average annual natural background for a generic site.
- No single component failure leads to very large consequences and no single event can simultaneously damage the multiple confinement barriers provided.

The analyses and assessments completed with the involvement of the Home Teams experts offer a well-developed technical basis for regulatory applications in potential host countries.

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